USEC
A Global Energy Company

June 9, 2006
AET 06-0071

Mr. Jack R. Strosnider
Director, Office of Nuclear Material Safety and Safeguards
Attention: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

American Centrifuge Plant
Docket Number 70-7004
Submittal of Planned Changes to the License Application for the American Centrifuge Plant (TAC Nos. L32306, L32307, and L32308)

Dear Mr. Strosnider:

Pursuant to requests from the U.S. Nuclear Regulatory Commission (NRC) staff on May 26 and June 6, 2006, USEC Inc. hereby submits planned changes for the License Application for the American Centrifuge Plant as Enclosure 1 of this letter. These planned changes will be finalized and submitted to the NRC in the next revision of the License Application and supporting documents. Enclosure 2 provides a revised response for Request for Additional Information Question NC-19.

If you have any questions regarding this matter, please contact Peter J. Miner at (301) 564-3470.

Sincerely,

Steven A. Toelle
Director, Regulatory Affairs

cc: S. Echols, NRC HQ
T. Johnson, NRC HQ
B. Smith, NRC HQ
C. Tripp, NRC HQ

Enclosure: As Stated

USEC Inc.
6903 Rockledge Drive, Bethesda, MD 20817-1818
Telephone 301-564-3200 Fax 301-564-3201 http://www.usec.com

NMSS01

Enclosure 1 of AET 06-0071

Planned Changes for the License Application for the American Centrifuge Plant

USEC complies with requirements for generators of hazardous and mixed waste. The State of Ohio has adopted a federal conditional exemption from the hazardous waste rules that is available under 40 CFR Part 266, Subpart N (OAC 3745-266).

1.1.7 Roadways

Two major four-lane highways service the DOE reservation: U.S. Route 23, traversing north-south, and U.S. 32/124, traversing east-west. The reservation is situated approximately three and one half miles from the intersection of U.S. Route 23 and U.S. 32/124. Ingress and egress from the reservation to these major roadways is by the Main Access Road, which connects to U.S. Route 23. The Main Access Road connects to the Perimeter Road, which encircles the fenced portion of the DOE reservation. Alternative ingress and egress from the reservation can be established from the north access road in the event of significant Main Access Road repairs. Service roads throughout the reservation connect to the Perimeter Road with access to the ACP controlled through security portals. The reservation roadways are depicted in Figures 1.1-1 and 1.1-2 (located in Appendix B).

1.1.8 Transition from Lead Cascade Demonstration Facility Activities to American Centrifuge Plant Activities

On February 24, 2004, the NRC granted USEC a license to possess and use source and special nuclear material at the American Centrifuge Lead Cascade Demonstration Facility (Lead Cascade) located on the DOE reservation in Piketon, Ohio. The Lead Cascade's license authorizes operation for a period of five years, which expires on February 24, 2009.

Depending on a number of factors, including cost and schedule, one of the following four options would be utilized to transition activities from the Lead Cascade possession and use license to the construction and operation license of the ACP.

1.1.8.1 Option 1: Subsume Lead Cascade Operations under the ACP

This option presumes that USEC would operate the centrifuge machines that comprise the Lead Cascade after February 24, 2009, the Lead Cascade license expiration date. USEC would terminate its possession and use license and transfer any remaining demonstration activities of the Lead Cascade to an authorized use within the ACP License. This would occur prior to February 24, 2009. The Lead Cascade facility descriptions would be reviewed to identify any potential changes to ACP facility descriptions and the changes would be evaluated in accordance with 10 CFR 70.72 and 70.32. USEC would notify the NRC well in advance of the transition of the Lead Cascade to the ACP. At that time, USEC would request a License Amendment and submit a more detailed Lead Cascade transition plan to NRC in accordance with the requirements of 10 CFR 70.38 and 10 CFR 40.42 for NRC review and approval.

The Lead Cascade UF_6 inventory would be transferred to the ACP prior to the license expiration date. USEC expects that most of the Lead Cascade centrifuge machines and equipment/components (i.e., piping, valves, other support system/components, etc.) will be used in the ACP. The re-use, refurbishment, or other disposition of the machines and system components will be based upon engineering evaluations and ACP design requirements. To the

extent Lead Cascade equipment is used as part of the ACP, decommissioning of that equipment will not be necessary. Equipment not utilized in the ACP will be handled in accordance with the requirements of 10 CFR 70.38 and 10 CFR 40.42.

1.1.8.2 Option 2: Renewal of Lead Cascade Demonstration Facility Possession and Use License

This option presumes that USEC would renew the Lead Cascade license in accordance with 10 CFR 70.73 and continue to operate the Lead Cascade concurrently with the activities being conducted under the ACP license. When NRC grants permission to operate the ACP, USEC would either terminate its possession and use license and transfer any remaining demonstration activities of the Lead Cascade to an authorized use within the ACP License as described in Option 1, continue to operate the Lead Cascade under its license for a period of time, or terminate its license in accordance with Option 3.

1.1.8.3 Option 3: Termination of Lead Cascade Operations

This option presumes that USEC would allow the Lead Cascade license to expire on February 24, 2009, the Lead Cascade license expiration date. The Lead Cascade UF_6 inventory would be transferred to an entity authorized to possess the material prior to the license expiration date. USEC expects that most of the Lead Cascade centrifuge machines and equipment/components (i.e., piping, valves, other support system/components, etc.) will be used in the ACP. The re-use, refurbishment, or other disposition of the machines and system components will be based upon engineering evaluations and ACP design requirements. To the extent Lead Cascade equipment is used as part of the ACP, decommissioning of that equipment will not be necessary. The Lead Cascade facility descriptions would be reviewed to identify any potential changes to ACP facility descriptions and the changes would be evaluated in accordance with 10 CFR 70.72 and 70.32. Equipment not utilized in the ACP will be handled in accordance with the requirements of 10 CFR 70.38 and 10 CFR 40.42.

USEC would notify the NRC well in advance of the license expiration date of its plans to execute this option. At that time USEC would submit a more detailed Lead Cascade license termination plan to NRC in accordance with the requirements of 10 CFR 70.38 and 10 CFR 40.42 for NRC review and approval.

1.1.8.4 Option 4: Phased Deployment

This option presumes that upon receipt of a license for the ACP, USEC would implement the initial phase of its commercial operations as described in Appendix C. A more detailed description may be found in document LA-3605-0003A, Addendum 1 of the ISA Summary. Thereafter, USEC would construct and install machines in phases until it reaches a capacity of 3.5 million SWU approximately four years after receipt of a license.

USEC would notify the NRC well in advance of the transition of the Lead Cascade to the ACP. At that time, USEC would request a License Amendment and submit a more detailed Lead Cascade transition plan to NRC in accordance with the requirements of 10 CFR 70.38 and 10 CFR 40.42 for NRC review and approval.

Section 8.6.1 A battery system maintenance procedure will be developed in accordance with existing plant battery system practices based upon nearly 50 years operating experience and the battery system manufacturer's recommendations. It is anticipated that general battery system inspections will be performed monthly in accordance with Table 8-1.

Section 8.9 USEC does not commit to all of the standards listed in this section.

Sections 10.4 a.) thru c.) The UPS final factory testing steps will be based upon the capacity (size) of the system, the precise type of batteries, the system configuration, and the intended function of the installed system.

Section 10.9 USEC does not commit to all of the standards listed in this section.

For the reference to this standard see Sections 2.6.4 and 2.6.7 of the ISA Summary for the ACP.

- *IEEE 484-2002, IEEE Recommended Practice for Installation Design and Installation of Vented Lead-Acid Batteries for Stationary Applications*

  USEC will satisfy the provisions of this standard.

  For the reference to this standard see Section 3.8.4 of the ISA Summary for the ACP.

- IEEE 603-1998, *Standard Criteria for Safety Systems for Nuclear Power Generating Stations*

  USEC commits to utilizing IEEE 603 Clauses 1 (Scope), 3 (Definitions) and 7 (Execute Features) and portions of Clauses 5 (Safety System Criteria), 6 (Sense and Command Features), and 8 (Power Source Requirements).

  USEC takes exception to the contents of IEEE 603 Clauses 2 (References), 4 (Safety System Design Basis), and Annexes A, B, and C. These clauses are not considered to be applicable or necessary due to their nuclear reactor content and redundancy with other IEEE standards and USEC's ISA. Annexes A, B, and C provide only "informative" details and references. In addition, USEC takes exception to portions of contents in IEEE 603 Clauses 5, 6, and 8 for the following reasons:

  Sections 5 and 5.1 Single-failure criterion will be applied only where needed to provide the reliability of the IROFS credited in the ISA.

  Sections 5.3

Should the addition of new processes or other changes to the ACP be necessary, evaluations of appropriate complexity for each process will be performed in accordance with 10 CFR 70.72, using established ISA methods to ensure the processes can be carried out in a manner such that compliance with the performance requirements of 10 CFR 70.61 are maintained. The ISA methods utilized for the ACP are described in section 3.1.2.1 of this License Application.

USEC maintains the ISA and ISA Summary so that it is accurate and up-to-date by means of a suitable configuration management system, described in Section 11.1 of this license application. ACP procedures specify the criteria for changing the ISA Summary. Changes to the ACP are evaluated against the ISA and ISA Summary using a change process that meets the requirements of 10 CFR 70.72. Changes to the ISA Summary are submitted to the NRC in accordance with 10 CFR 70.72(d)(1) and (3). USEC will provide to the Commission, 180-days prior to the introduction of UF_6 in the American Centrifuge Plant, a revised ISA Summary that incorporates all changes that have occurred since the issuance of the materials license. The ISA accounts for any changes made to the ACP or its processes (e.g., changes to the site, operating procedures, or control systems). Any facility change, operational change, or change in the process safety information that may alter the parameters of an accident sequence is evaluated by means of the ISA methods. USEC evaluates proposed changes to the ACP or its operations by means of the ISA methods and designates new or additional IROFS, along with appropriate management measures, as necessary. USEC will periodically review IROFS per the requirements of 10 CFR 70.62(a)(3) to ensure their availability and reliability for use, and consistency with the ISA. As the final design is developed for the ACP, the management system and design approach will require that the final designs be reviewed against the ISA to ensure the ISA accurately reflects the ACP design and operations, identifies the credible accident sequences and appropriate assumptions, and credits the IROFS necessary to meet the performance requirements of 10 CFR 70.61. The license should be conditioned as follows: Upon completion of the design and updating of the ISA and ISA Summary, USEC shall provide the Commission with 120 days advance notice of its plan to introduce UF_6 in the American Centrifuge Plant in order to conduct its inspections involving the ISA and ISA Summary that are required by 10 CFR 70.32(k).

USEC also evaluates the adequacy of existing IROFS and associated management measures and makes any required changes to the ACP and/or its processes. If a proposed change results in a new type of accident sequence (e.g., different initiating event or significant changes in the consequences) or increases the consequences and/or likelihood of a previously analyzed accident sequence within the context of 10 CFR 70.61, USEC evaluates whether changes to existing IROFS and associated management measures are required, or if new IROFS or management measures are required. For any changes that require prior NRC approval under 10 CFR 70.72, USEC will submit an amendment request in accordance with 10 CFR 70.34 and 70.65.

The Engineering Manager is responsible for maintaining the ISA and ISA Summary (i.e., reviewing proposed changes, performing analyses, and ensuring implementation of required updates). The Regulatory Manager is responsible for submitting the required changes to the NRC and coordinating information requests from the NRC.

3.5 Integrated Safety Analysis Maintenance

As stated previously, the ISA is a compilation of the design and analysis documentation utilized to identify the potential accident sequences that could occur, designate IROFS to either prevent such accidents or mitigate their consequences to an acceptable level, and identify the management measures to provide reasonable assurance of the availability and reliability of IROFS. The ISA Summary is a synopsis of the ISA and contains the information required by 10 CFR 70.65(b). The ISA Summary is updated to reflect changes to the ISA.

The ISA accounts for any changes made to the ACP facilities or its operations are evaluated in accordance with the requirements of the 10 CFR 70.72 change process. Any facility change, operational change, or change in the process safety information that may alter the parameters of an accident sequence is evaluated by means of the ISA methods. USEC periodically reviews IROFS per the requirements of 10 CFR 70.62(a)(3) to ensure their availability and reliability for use and consistency with the ISA. USEC evaluates whether changes to existing IROFS and associated management measures are required, or if new IROFS or management measures are required. The bases (including assumptions and initial conditions) for the ISA are maintained and controlled via the various management measures identified in Chapter 11.0 of this license application. This includes, but is not limited to the preventive maintenance, corrective action, configuration management, and audit/assessment programs.

For any changes to the accident sequences in the ISA, or the addition of any new accident sequences to the ISA, USEC shall address and document the following considerations in the ISA: (1) The accident sequence will specify whether the event is characterized by a frequency of occurrence or by a probability of failure on demand, and will perform all necessary mathematical operations appropriate to the type of event. (2) The accident sequence will distinguish between frequencies and probabilities applicable to a single item and those applicable to a population of identical items; (3) The accident sequence will take demand rates into consideration, for all items characterized by a failure on demand; (4) The applicant will justify independence for any combination of repeated events, or else reduce the assigned likelihood of the combined failure to conservatively bound common-mode failures; and (5) For criticality accident sequences, the accident sequence will consider whether less reactive physical conditions could lead to a higher likelihood of criticality.

3.6 References

1. LA-3605-0003, Integrated Safety Analysis Summary for the American Centrifuge Plant

2. NUREG-1513, *Integrated Safety Analysis Guidance Document*, U. S. Nuclear Regulatory Commission, Office of Nuclear Material Safety and Safeguards, Washington, DC, May 2001

3. NUREG-1520, *Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility*, U. S. Nuclear Regulatory Commission, Office of Nuclear Material Safety and Safeguards, Washington, DC, January 2002

- Procurement controls are provided; and
- Receipt inspection is used when specified.

Components and features that are identified in the NCSEs or the ISA are analyzed to determine the "boundary" of the system, encompassing those interconnecting and/or supporting items that are essential to ensure availability and reliability. The boundaries are identified on system drawings, and the configuration is verified to be as-built. These components and features are maintained in a design control document for the building or process. Each time a change is planned, the document is reviewed by the individual (e.g., design authority, systems engineer, operations manager, maintenance, etc.) planning the change to determine if the change affects an IROFS or double contingency control. Changes that could establish new fissile material operations or affect established fissile material operations are reviewed by NCS. The NCS Program establishes and maintains NCS safety limits and NCS operating limits for IROFS and double contingency controls in nuclear processes and maintains adequate management measures to ensure the availability and reliability of the IROFS and the double contingency controls.

The change control process specifies the organizations required to perform reviews of changes. If an item is relied on for the criticality safety of an operation (i.e., is an IROFS or a double contingency control), it will be identified and NCS reviews the NCSE for the specific operation and determines if the change affects the analysis performed and the conclusions made in the NCSE. The change request will be approved by NCS only if the change does not adversely impact NCS, or once a revised NCSE has determined that the change is acceptable and meets NCS Program requirements. If a change affects the ISA Summary, it is updated appropriately. In this way, modifications to controlled operations are evaluated and approved prior to implementation and placing the affected structures, systems, or components in service.

Records management and document control (RMDC) is another element of CM and is described in Section 11.7 of this license application. Procedures, documents, and records control programs provide for centralized control and issuance of documents essential to the maintenance of the design history, and a repository for records to verify this maintenance. NCSEs are specifically included in the index of documents that are required to be controlled.

5.3.4 Operation Surveillance and Assessment

To ensure that the NCS Program is properly established and implemented, walk-throughs, assessments, and audits are utilized.

Operating SNM process areas are reviewed on a regular basis through a combination of walk-throughs and reviews by work crew supervision. NCS walk-throughs of facilities that may contain fissile material operations are performed by NCS personnel to determine the adequacy of implementation of NCS requirements and to verify that conditions have not been altered to adversely affect NCS. These walk-throughs are performed as specified by the NCS procedure on walk-throughs. For example, a walk-through inspection can be performed in response to trend data, at the request of the operations personnel, or due to concerns raised by employees or NCS personnel. As a

within specified values. If two controls are implemented for one parameter, the violations or failure scenarios addressed by the controls will be independent. Application of this principle ensures that no single credible event can result in an accidental criticality or that the occurrence of events necessary to result in a criticality is not credible.

The NCSE will document the basis for the conclusion that a change in a process or parameter is "unlikely." The basis may be an engineered feature, administrative control, the natural or credible course of events, or any combination of these or other means necessary to ensure the change is unlikely to occur. The parameters or conditions relied on and the limits must be specified and justified in the NCSE and controlled. Management measures described in Chapter 11.0 of this license application and other safety programs are sometimes used to help ensure a change in a process or parameter is "unlikely." For example, the Radiation Safety Program and/or the Fundamental Nuclear Material Control Plan may be credited with providing controls on fissile material handling; the Fire Safety Program may be credited with providing controls on combustible material loading and/or hot work activities in fissile material processing/storage areas; the Procedures Program may be credited with ensuring compliance with procedures; etc.

Where the natural or credible course of events is relied upon in whole or in part to prevent a process condition change, no specific additional controls will be necessary to maintain them. The factors that influence the process are described in sufficient detail in the NCSE as items related to NCS and programmatically controlled. For items that are established, maintained, and implemented by non-NCS programs, credit for availability and reliability is established as described in Section 11.1 of this license application without the need for additional NCS controls. For situations where the NCS-credited controls do not provide adequate assurance of availability or reliability (i.e., situations where non-NCS programmatic and physical plant changes could adversely affect the intended criticality safety function of the items relied upon for criticality safety), specific NCS controls are established, maintained, and implemented to ensure criticality safety.

Use of the natural and credible course of events or other means in lieu of specific administrative or engineered controls for double contingency protection requires prior NRC review and approval. The request for review and approval will include a justification of why administrative or engineered controls are not needed, a description of the proposed measures in sufficient detail to permit an understanding of their safety function, and a justification of their inherent unlikeliness. This requirement does not apply to NCS reliance on the proper implementation of other plant programs or management measures that are described in Chapter 11.0 of this license application. This requirement also does not apply to accident sequences determined to be non-credible or those sequences which do not result in a critical configuration even with the loss of both double contingency controls.

The NCS evaluation process involves a review of the proposed operation and procedures or work instructions, discussions with the subject matter experts to determine the credible process upsets which need to be considered, development of the controls necessary to meet the double contingency principle, and identification of the assumptions and equipment (i.e., physical controls) needed to ensure criticality safety.

Emergencies arising from unforeseen circumstances can present the need for immediate action. If NCS expertise or guidance is needed immediately to avert the potential for a criticality accident, direction will be provided orally or in writing. Such direction can include a stop work order or other appropriate instructions. Documentation will be prepared within 48 hours after the emergency condition has been stabilized.

New operations must comply with the double contingency principle.

5.4.2.1 Non-Fissile Material Operations

Some operations involve situations in which the uranium has an enrichment of less than 1 wt. percent ^{235}U or an inventory of less than 100 g ^{235}U. These operations are termed "non-fissile material operations" and are performed without the need for NCS double contingency controls. The determination of which operations are fissile versus which operations are non-fissile may be contained within a NCSE or as a separate document. When the determination is outside a NCSE, the determination need not be performed by a qualified NCS Engineer. The determination of an operation being non-fissile must include normal and credible abnormal upset conditions to ensure the enrichment and/or inventory are maintained below 1 wt. percent ^{235}U or below 100 g ^{235}U. Controls are sometimes applied to a non-fissile material operation to ensure it does not inadvertently involve fissile material. These controls can be either engineered or administrative and will be incorporated into applicable operating procedures or work instructions when it is determined they are needed to maintain the non-fissile material operation below either 100 g ^{235}U or 1 wt. percent ^{235}U. This determination is made by the responsible line manager.

5.4.3 Design Philosophy and Review

Through the CM Program, designs of new fissile material equipment and processes must be approved by NCS before implementation. Where practical, the use of engineered controls on mass, geometry, moderation, volume, concentration, interaction, or neutron absorption will be used as the preferred approach over the use of administrative controls. Advantage will be taken of the nuclear and physical characteristics of process equipment and materials, provided control is exercised to maintain them if they may credibly degrade such that control of the parameter is jeopardized.

The preferred design approach includes two goals. The first is to design equipment such that NCS is independent of the amount of internal moderation or fissile concentrations, the degree of interspersed moderation between units, or the thickness of reflectors. The second is to minimize the possibility of accumulating fissile material in inaccessible locations and, where practical, to use favorable geometry for those inaccessible locations. Passive design controls are preferred to active design controls. The adherence to this approach is determined during the preparation and technical review of the NCSE performed to support the equipment design. This preferred design approach is implemented as described in NCS procedures. Deviations from the preferred design approach are justified in supporting documentation to the NCSEs.

To protect against the loss of coverage, the CAAS includes redundant decision logic, a backup power supply, detector status information and system self-diagnostic information are provided to the X-3012 building ACR and X-1020 building. The CAAS has been designed to survive and/or withstand credible abnormal events as described in the accident analysis for a sufficient time to warn personnel to evacuate. In the event CAAS coverage is lost for an operation, plant procedures provide for compensatory actions, which may include shutdown of equipment, limiting access, halting movement of uranium-bearing material, or other actions.

Additional information provided by the CAAS includes a historical log of events and the capability to monitor and record the criticality accident for managing the post-accident situation and any remedial action. Nuclear accident planning and response is discussed in Section 2.2.4 of the Emergency Plan for the American Centrifuge Plant.

5.4.4.1 Portable CAAS

In the event a fissile material operation requiring CAAS coverage is performed beyond the detection range of established CAAS instrumentation, a portable unit may be used. The portable unit has the same detection capabilities as the permanently installed units, although those capabilities may be based on gamma radiation. Alarm annunciation, however, is usually limited to the immediate area within the audible range (confirmed to 65 feet or more) of the unit's alarm with an additional telemetric link to the X-3012 ACR and X-1020. This link will transmit the location of the unit, if mobile, and allow the use of the plant PA system to warn personnel within 200 feet of the area of the portable unit to evacuate. A portable unit will not be used for more than 24 continuous hours and it may be located indoors, outdoors, or on a vehicle.

If fissile material operations in an area without a permanently installed CAAS are required to exceed 24 continuous hours, all personnel not directly involved in the affected operations, or otherwise required for the safety or security of the facility, will be evacuated from an area within a 65 foot radius of the fissile material until the operations are concluded. In addition, affected operations shall be terminated as soon as safely achievable.

5.4.5 Technical Practices

5.4.5.1 Application of Parameters

Moderation

Water is considered to be the most efficient moderator commonly found in the ACP. This is because optimally moderated UO_2F_2/water solutions are more reactive than hydrocarbon oil/UF_4 solutions at worst credible concentrations experienced in vacuum pumps (Reference 13). When moderation is not controlled either optimum moderation or worst credible moderation is assumed as the normal case when performing analyses. When moderation is controlled, credible abnormal process upset conditions determine the worst-case moderated conditions. Generally, moderation control is not maintained by measurement; however, when used, dual independent sampling methods are implemented.

Computer Calculations

For those cases where adequate references are not available, NCS computational analyses are performed, which involve the calculation of k_{eff} to determine whether the system will be subcritical under both normal and credible abnormal process conditions. Computer codes that simulate the behavior of neutrons in a process system or that solve the Boltzmann transport equation are used.

Computer calculations of k_{eff} provide a method to relate analytical models of specific system configurations to experimental data derived from critical experiments. A critical experiment is defined as a system that is intentionally constructed to achieve a self-sustaining neutron chain reaction or criticality. Critical experiments that have specific, well-defined parametric values and are adequately documented are termed benchmark experiments. Computer codes are validated using experimental data from benchmark experiments that, ideally, have geometries and material compositions similar to the systems being modeled.

Validation of the computer code determines its calculational bias or uncertainty as well as the effective margin of subcriticality. The validation involves the modeling of benchmark critical experiments over a range of applicability. Because the k_{eff} value of a critical experiment is essentially 1, the bias of the code is taken to be the deviation of the calculated values of k_{eff} from unity. Statistical analysis is employed to estimate the calculational bias, which includes the uncertainty in the bias and uncertainties due to extensions of the area of applicability, as well as the effective margin of subcriticality. Uncertainty in the bias is a measure of both the precision of the calculations and the accuracy of the experimental data. The validation of the computer code specifically defines the maximum acceptable k_{eff} used to determine subcriticality.

The margin of subcriticality used for the plant results in a k_{eff} upper safety limit that ensures that there is a 95 percent confidence that 99.9 percent of future k_{eff} values less than this limit will be subcritical. A minimum margin of subcriticality of 0.02 in k_{eff} is used to establish the acceptance criteria (i.e., upper safety limit) for criticality calculations for abnormal conditions at 5 percent ^{235}U enrichment and below. Above 5 percent ^{235}U enrichment, a minimum margin of subcriticality of 0.05 in k_{eff} is used. Also, for normal case calculations supporting processes that are not under moderation control, a minimum margin of subcriticality of 0.05 in k_{eff} is used. Abnormal conditions are changes to a controlled parameter that result in a violation of the limit on that parameter. For example, in an operation that relies on maintaining spacing between fissile units, an error that results in the units being closer than the limit would represent an abnormal condition. Similarly, operations that rely on moderation control of UF_6 would be in an abnormal condition when the moderation control was lost and operations that rely on control of ^{235}U mass would be in an abnormal condition when the mass limit was violated.

The upper safety limit varies with the computer system, codes, cross sections, and materials used in the validation.

The calculation of k_{eff} is accomplished by the use of computer codes that utilize Monte Carlo techniques to determine k_{eff} of a system. Computer models representing the geometrical configuration and material compositions of the system are developed for use within the code. The development of appropriate models must account for or conservatively bound both normal and

Prior to implementing changes to processes based on calculations requiring extension to the validated area of applicability as described above, NRC review and approval shall be obtained. The request for NRC review and approval shall include a description of the change, the reason that such a change is needed, and the method used to extend the area of applicability. The only exception to this requirement is for processes similar to those evaluated for the initial license application having the fissile material in the form of UF_6 below a moderation level of an H/U of 8.

The methodology used in a validation report involves statistical analysis to determine the bias and bias uncertainty for the critical experiments included in the validation. Guidance from NUREG/CR-6698, *Guide for Validation of Nuclear Criticality Safety Calculational Methodology*, is used to perform the validation. The upper safety limit is computed by subtracting the absolute value of the bias, the bias uncertainty, and the minimum margin of subcriticality from unity. Positive bias is not credited. The exact statistical technique used to obtain the bias and bias uncertainty depends on the specific validation report. The techniques used in Reference 11 included the lower tolerance limit or the lower tolerance band for normally distributed data and a non-parametric technique for non-normally distributed data.

The computer codes and cross sections used in performing k_{eff} calculations are maintained in accordance with a configuration control plan. Quarterly, or prior to use, one of the following is performed: a bit-by-bit comparison of the production version of the software (executable modules and data libraries) versus an archived production version; or a comparison of the output from all validation cases versus archived output of all validation cases from the original validation performed when the production version was installed to ensure no changes in the calculated k_{eff} for the validation cases.

Changes to the hardware or software are evaluated in accordance with 10 CFR 70.72 change requirements. Some changes are expected to result in changes to the calculational algorithm and will require a new validation. Such changes include revisions to the software used to calculate reactivity, updates to the cross section libraries, changes to the operating system kernel, changes to the central processing unit, or changes to the motherboard. Other changes are not expected to result in changes to the calculational algorithm and will require only that the validation cases be re-run and compared to the original results. Such changes include increasing the available RAM, changing a hard drive, graphics card, network interface card, or other peripheral. In the Microsoft Windows environment, periodic changes to components of the operating system are common as Microsoft issues updates or patches to the platform. Also, installation and modification of software not used to calculate reactivity will be performed to support day-to-day business needs. These minor changes are not expected to impact any reactivity calculations, but to ensure this, a verification of the validation cases will be performed at least quarterly as described above.

The System Administrator, a NCS engineer, is responsible for controlling access to the software.

In accordance with the requirements of 10 CFR 70.72, the ACP implements change control processes for changes to the physical plant and for changes to procedures and controlled documents. These processes are described in Sections 11.1.4.1 and 11.1.4.2 of this license application, respectively. The Plant Safety Review Committee reviews appropriate changes to the ACP or to ACP operations, including tests and experiments, as specified in procedures. Procedures also specify the approval authority for the changes.

11.1.4.1 Control of Changes to the Physical Plant

The ACP has implemented a change control process using written procedures to control changes to the physical plant. This change control process meets the requirements established in 10 CFR 70.72 and in the QAPD. Key elements of the change control process are described in the following paragraphs:

- Requests for engineering assistance, after initiator's management approval, are forwarded to the DA for:
  - ➤ Review to determine if the proposed change is acceptable based upon scope, applicability, justification, and/or technical merit;
  - ➤ Engineering approval; and
  - ➤ Disposition and assignment to the appropriate Engineering discipline.
- Construction Project requests for plant modifications, additions, or changes have a 10 CFR 70.72 review performed to determine if the change can be made without prior NRC approval. Information utilized in the 10 CFR 70.72 review includes the following, as appropriate:
  - ➤ SRDs;
  - ➤ Conceptual design descriptions;
  - ➤ Drawings/specifications; and
  - ➤ Other documentation providing a project description.
- Modifications (permanent and temporary) are evaluated, as appropriate, for any required changes or additions to the plant's procedures, personnel training, testing programs, or the ISA Summary. Modifications are also evaluated, as appropriate, for potential radiation exposure, potential chemical exposure, and worker safety requirements and/or restrictions. Other areas of consideration in evaluating modifications may include: modification costs, similar completed modifications, QA aspects, potential equipment availability or maintainability concerns, constructability concerns, environmental considerations, and human factors. Modifications that

establish new fissile material operations or affect existing fissile material operations are evaluated by nuclear criticality safety (NCS).

- Critical repair parts for IROFS are identified during the design process.
- Proposed plant changes receive an independent, technical review that considers the technical feasibility and merit of the proposed change and the identification of appropriate interfaces for inclusion in the change package (e.g., procedures, training, safety).

A final review prior to release for operation is conducted which verifies that:

- The safety analysis documentation is complete and approved
- Operational procedure changes, if required, are completed and other supporting procedure changes have been initiated
- Operational training and qualification changes, if required, have been completed
- Design changes are completed and any as-built changes are identified and approved
- Document changes, if required, are completed
- For temporary changes, the change duration is documented and the modified equipment tagged
- Post-modification testing has been successfully completed
- Appropriate approvals have been obtained

11.1.4.2 Control of Changes to Procedures and Controlled Documents

Changes to procedures and controlled documents are controlled in accordance with the programs described in Sections 11.4 and 11.7 of this license application, respectively.

11.1.5 Assessments

The CM Assessment Program systematically evaluates the development and effective implementation of the CM Program processes. It assesses the adequacy of the implementation of administrative requirements, the configuration of items, and their documentation. The CM Assessment Program includes both initial and periodic assessments. Both document assessments and physical assessments (system walk downs) are conducted periodically to confirm the adequacy of the CM function.

procedure format and style and that it is consistent with the procedure-writing guide. Verification consists of a walk-down of the procedure in the field or a tabletop walk-through. A standard checklist is used to ensure required attributes are included.

11.4.2.4 Validation

The purpose of procedure validation is to ensure that no technical errors or human factor issues were inadvertently introduced during the procedure review process. Validation is required for new procedures or for intent changes to the procedure. Validation is performed in the field by qualified personnel, and may be accomplished by detailed scrutiny of the procedure as part of a walk-through exercise or as part of a walk-through drill (particularly for emergency or off-normal procedures). If the particular system or process is not available for a walk-through validation, talk-through may be performed in the particular shop or training environment. Performance of procedure validation is documented.

11.4.2.5 Review

Drafts of new procedures and procedure changes are distributed for technical reviews, safety discipline reviews (e.g., nuclear criticality, fire, radiation, industrial, and chemical process safety), and cross-discipline reviews, as needed. Nuclear criticality safety reviews drafts of new procedures and procedure changes that could affect fissile material operations.

Functional area and cross-discipline reviews are performed for the new procedure or procedure change. Comments/questions generated during the review process are resolved with the originating organizations. 10 CFR 70.72 and intent/non-intent screenings are performed for new and changed procedures (except minor administrative changes that are processed according to the procedure process).

Any new or revised NRC requirements that are promulgated are evaluated to determine the impact on existing implementing procedures or to identify the need for new implementing procedures. Procedures are reviewed following unusual incidents; such as an accident, unexpected transient, significant operator error, or equipment malfunction to determine if changes are appropriate based on the cause and corrective action determination for the particular incident. Procedure changes that are necessary because of a system modification are addressed in Section 11.1 of this license application, as part of the modification control process.

In addition, the Plant Safety Review Committee will review:

- Each new procedure required by Section 11.4.2.1 for this license application
- Each proposed change to procedures required by Section 11.4.2.1 of this license application, if the proposed change constitutes an intent change (i.e., a change in scope, method, or acceptance criteria that has safety significance)

Enclosure 2 of AET 06-0071

Revised Response for RAI Question NC-19

Enclosure 2 of AET 06-0071

NRC's RAI NC-19 requested the applicant to clarify whether it would document exceptions to the preferred design philosophy in license application Section 5.4.3. The applicant responded that it did not intend to document the reasons for not following the preferred approach, such as when relying on administrative controls.

The NRC staff indicated that, although it did not expect that engineered controls could be used in every instance, there should be a good reason for deviating from the preferred design philosophy and this reason should be documented.

The applicant stated that administrative controls were used mainly in maintenance operations, in which use of passive or active engineered controls is impractical due to the hand-on nature of these operations. Examples of other operations involving administrative controls are the vacuuming of spills, container handling, and equipment transport. The requirement to ensure that criticality is highly unlikely under 10 CFR 70.61 drove the vast majority of ACP processes to rely on engineered controls. The NRC staff agree that the design of the majority of the ACP facility relies extensively on inherently safe, passive equipment design, consistent with the applicant's commitment to the preferred design philosophy. As with other facility changes, replacement of an engineered control with an administrative control would require NRC pre-approval under 10 CFR 70.72. No further response was necessary beyond a written confirmation of USEC's statements made during the meeting as reflected above.

USEC Revised Response

The use of the preferred design approach with regard to nuclear criticality safety is described in Section 5.4.3 of the License Application. USEC has included commitments to use the preferred design approach, with a preference for passive design controls over active design controls. Additionally, USEC has added the following statement to this section: "Deviations from the preferred design approach are justified in supporting documentation to the NCSEs."

This response supersedes USEC response provided by AET 06-0038 dated May 23, 2005.